BEST COPY
AVAILABLE

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/766
UKRAINIAN SSR
February 1964
Original: RUSSIAN

Confidential until official release during Conference

Abstract

DETERMINATION OF INTENSITY OF THE RADIOACTIVE CONTAMINATION IN THE OCEAN BASED ON THE NEW EXCHANGE DATA

V.I. Belyaev, A.G. Kolesnikov and B.A. Nelepo
Marine Hydrophysical Institute of the A.S. UkSSR, Sevastopol

Till now the term "age" was used to determine spreading rate of the radioactive contamination in the ocean. Numerous authors considered "the age" of the Atlantic Ocean to be 500 years. However, one cannot mechanically transfer the term "age" from other sciences to the ocean. The obtained values will not be reliable because of the complication of observed processes. This often leads to the incorrect conclusion of the low exchange intensity in the ocean and consequently to underestimation of contamination danger of the vast ocean areas caused by radioactive wastes.

Later some investigators used diffusion coefficient values derived from the experimental data. And though calculations of the contamination spreading rate were of great interest they gave us only general ideas about the process above-mentioned. This can be explained by the fact that up to this time the simplified mathematical concepts and models were used (not considering some peculiarities such as the vertical component of the average current velocities and real law of the radioactive fall-out on the ocean surface).

The paper includes the results of spreading rate calculations of the radionuclides with various half-lives (based on numerical solution by computers) of the complete equation of turbulent diffusion. This involves the really observed value of fall-out on the surface from the beginning of the first tests. Turbulent diffusion coefficients and vertical components of the average current velocities were calculated from the data collected aboard r/v "Mikhail Lomonosov" in the Atlantic Ocean.

Translation provided by courtesy of the submitting Government

Obtained solutions of the radionuclide concentration and the radioactivity at various depths in the Atlantic Ocean were analysed.

The following conclusion was drawn:

(1) Depth of penetration of the radioactive isotopes increases with the increase of the radioactive isotopes half-life.

(2) The mode of distribution of the radioactive elements generally repeats distribution of density, indicating the importance of thermocline.

(3) Calculated curves for Sr^{90} and Cs^{137} distribution with depth are in complete agreement with the observed data.

(4) Analysis of the obtained solutions leads to much higher spreading rate of the radioactive contamination than existing knowledge. The latter is in good agreement with the observation data of Sr^{90} concentration distribution with depth carried out by r/v "Mikhail Lomonosov".

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/767
AUSTRIA
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Experiments on the Reprocessing of Uranium Fission Product Solutions by the Exclusive Use of Ion Exchangers.

O. Bobleter
Atominstitut der Österreichischen Hochschulen, Vienna

In the reprocessing of nuclear fuel material, ion exchangers have so far been employed for the seperation of certain elements only. The costliness of exchanger material as well as its instability in the case of high radiation doses have rendered its general use impracticable.

The development of inexpensive cation exchangers on wood basis was made possible by a Research Contract from the IAEA. These exchangers demonstrated good seperation properties for certain fission product elements, and it therefore seemed feasible to examine a complete reprocessing procedure using only ion exchangers.

The maximum loading factor and elution velocities possible for uranium and the main group of fission products were determined in several column experiments.

In the first exchange step, a pre-seperation of uranium from the fission products (Cs^{+}, Sr^{2+}, Y^{3+}, etc.) was carried out using an anion exchanger (Dowex 1 x 8), 4.2 N Aluminium nitrate solution gave good results as elution medium. However, not all elements may be separated from uranium by this first step. So part of cerium-III remains adsorbed with the uranium on the exchanger. In this case, the maximum elution velocities are relatively low.

In the second exchange step, the final separation of uranium from the fission products, which remain adsorbed with it on the anion exchanger, is carried out with a sulfitated cation exchanger on wood basis. The uranium-cerium-III separation can be successfully

obtained with a diluted NH_4NO_3-HNO_3 solution, and high elution velocities are possible, although both elements have similar nitrate complex properties.

In the third exchange step, the fission products are separated singly, also with a cation exchanger on wood basis. Good results were obtained by elution with nitric acid, the concentration of which was stepwise increased. Cs^+, Sr^{2+}, and Y^{3+}, which still contained $Al(NO_3)_3$ from the first exchange step, were separated by elution with 0.015, 0.14 and 0.47 M HNO_3. In this case also high elution velocities can be applied.

The beta and gamma doses were calculated for the exchanger which is irridiated to the greatest extent. For this calculation it was assumed that the daily fuel consumption of a 100 MW_t nuclear reactor has to be reprocessed by means of ion exchangers. The result of the calculation indicates that the process can be easily conducted in such a manner that no significant radiation damage occurs to the exchanger material.

The possibilities of reprocessing nuclear fuel by ion exchangers in small plants can be of advatage to many reactor staions. In addition to this, a complete separation of the fission products lessens the storage problem and gives better possibilities for the technical application of certain fission elements.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/768
ARGENTINA
February 1964
Submitted in:
SPANISH and FRENCH

Confidential until official release during Conference

Abstract

APPLICATION OF HEAP-LEACHING TO THE PROCESSING OF ARGENTINE ORES

Pedro N. Stipanicic, Rafael C. Coppa and
Aldo M. Cecchetto

The great area of Argentina and the scattered distribution thereon of numerous uranium ore bodies possessing special characteristics has made it necessary to study the possibility of applying heap-leaching as an economic method of ore-processing.

The need to attempt processing with unconventional methods arises from the presence of various unfavourable factors, such as:

(a) the existence of deposits of reasonable size but low grade, whose ores do not justify the cost of transport to distant processing centres and whose low uranium content makes the installation of a conventional processing plant an uneconomic proposition;

(b) the existence of many uranium ore bodies of medium, normal or even high grade, but so small in size that it would not be worth while setting up conventional concentrating plants, added to the considerable freight costs resulting from the distance to be covered to reach them;

(c) the existence of deposits from which normal grade ores can be obtained by selective working methods which, however, leave behind appreciable quantities of uranium of marginal or low grade, so that the deposits are not fully exploited.

The paper briefly reports the results of heap-leaching trials carried out with various Argentine uranium and copper-uranium ores, on both laboratory and semi-pilot-plant scale, more particular attention being devoted to the results for one typical ore, that from the "Don Otto" deposit (Salta Province).

The study for this special case includes:

1. general considerations regarding the method;
2. laboratory tests using 1-metre columns;
3. intermediate-stage tests using 4-metre columns;
4. results of treating an experimental heap of 700 tons of ore;
5. description of a production plant for the treatment of 36 000 tons of ore.

The method adopted can be summarized as follows:

The coarsely-milled ore is placed on a water-proofed platform tilted slightly towards one corner. The parallelepipedal heap of ore, with vertical sides maintained in place by a suitable surround, has a horizontal top surface. The leaching solution is poured over the surface of the heap and the percolation rate is controlled by a layer of clay of variable thickness on top of the ore. The solution circulates through the mass and collects at the level of the platform, where a pebble bed facilitates normal drainage, thence passing into a collecting tank. Here the pH of the solution - which has been partially neutralized during its passage through the ore - is adjusted and the solution is then re-cycled to the top of the heap. This cycle (percolation - pH adjustment - re-cycling - percolation) is repeated until the maximum quantity of uranium has been leached out of the ore, after which the rich solutions are removed by washing with acid water. The uranium-bearing liquids are then recovered by precipitation with milk of lime and, after drying, the lime preconcentrate is sent to a plant for final purification.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/769
ARGENTINA
February 1964
Original: SPANISH

Confidential until official release during Conference

Abstract

STUDIES ON FISSION PRODUCTS IN THE ENVIRONMENT

D. Beninson, E. Vander Elst and E. Ramos

An evaluation is presented on the transfer and fate of fission products released into the environment, using information resulting from fallout studies and from laboratory and field experiments related to waste disposal.

The paper presents data from radiochemical and -spectrometry on airborne activity, fallout rate and cumulative deposition in Argentina for the past seven years. Relationships between airborne activity, deposition and rainfall are studied, as well as seasonal trends and other effects.

Food chain contamination is examined with special emphasis given to pathways leading to man for strontium, cesium and iodine radionuclides. Correlations of food levels with fallout rates and deposition are attempted.

The soil-plant transfer step is studied using fallout data and information from experimental work relating to waste disposal.

Some attention is paid to fresh water food chains and to biological indicators which may be useful in monitoring the vicinity of nuclear installations.

The uptake of Strontium-90 by humans is specially considered. Using data of diets and bones of infants and children of different age groups, strontium to calcium discrimination factors are worked out and discussed together with those obtained from stable strontium measurements.

In an appendix, the paper includes data on radionuclide levels in materials from the Antarctic.

Translation provided by courtesy of the submitting Government

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/770
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

CONTRIBUTION OF THE REALIZATION OF A BIG NUCLEAR POWER PLANT IN THE DESIGN AND MANUFACTURE OF EQUIPMENT FOR REACTORS.

P. KLEES-Ingénieur
Chef de Section.

Ch. GERARD-Ingénieur
Chef de Service.

Division Nucléaire
Ateliers de Constructions
Electriques de Charleroi.

(BELGIQUE)

CHAPTER I - Introduction.

Thé proposed paper shows, on the basis of the experience accumulated during the " ARDENNES POWER PLANT (Givet - France) " construction, the progress made by the manufacturing industry in the refinement of the fabrication procedures and outlines the future trends.

CHAPTER II - Characteristic elements in the design of Systems equipments.

1. Functional requirements of a nuclear power plant.

This paragraph sums up the functional requirements of the ARDENNES Plant, which have a direct bearing on the systems construction and equipment specifications.

2. Peculiar characteristics in the systems design.

2.1. Auxiliary systems.

The hereabove mentioned requirements have led to special solutions concerning :

Translation provided by courtesy of the submitting Government

- corrosion control,
- regulation of soluble poison concentration and of water quality in relation with core operation,
- handling and treatment of radioactive charges and fluids,
- complete operational safety and high systems disponibility.

2.2. Control and Instrumentation.

This paragraph shows how have been resolved several control problems specificaly tied with the plant operating safety :

- design of a reliable power source for vital auxiliaries,
- analyse and preventive protection against accidents of nuclear origin,
- respect of the safety margins assigned by the core design,

2.3. Isolation philosophy for the reactor containment.

The isolation philosophy adopted for the ARDENNES Nuclear Plant is commented and conclusions are drawn to other plants.

3. Codes and Standards.

Typical examples are given which show the fundamental importance in the choice of codes and standards. This choice is all the more subject to discussions as are involved, like in the ARDENNES Plant, constructors and utilities of different countries.

4. Materials.

For the main parts of the primary installation, this paragraph describes the choice of the materials and the reasons for it.

CHAPTER III - Characteristic aspects in the equipment manufacture.

On the basis of practical experience obtained by the constructors of the ARDENNES Plant, this chapter defines new methods and procedures developped or industrialized for :

- welding and cladding,
- chroming,
- execution of small fabrication tolerances,
- heat treatments.

This chapter attemps to define what should be the criterias and the means adopted for :

- control of material and fabrication in shops,
- acceptance tests of the different equipment categories,
- nuclear cleanliness

CHAPTER IV - Specific test installations for nuclear equipment.

This chapter describes the installations especially designed for the development and testing, in operating conditions, of :

- primary pumps and valves,
- core elements,
- control rod drive mechanisms

CHAPTER V - Future trends in systems and equipments design, following the evolution of water power plant parameters.

CHAPTER VI - Conclusions.

The safety margins and fabrication procedures initialy adopted are evaluated, and from there on are proposed construction criterias industrialy adequate.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/771
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

REPROCESSING OF NUCLEAR FUELS BY VOLATILIZATION

J. Schmets
Head of the Chemical Engineering section of the Chemistry department

G. Camozzo
Secondant from the Research and Information division of Euratom, Brussels
Member of the Chemical Engineering section of the Chemistry department

A. Francesconi
Member of the Chemical Engineering section of the Chemistry department

P. Godrie
Member of the Chemical Engineering section of the Chemistry department

R. Heremans
Member of the Chemical Engineering section of the Chemistry department

G. Pierini
Secondant from the Research and Information division of Euratom, Brussels
Member of the Chemical Engineering section of the Chemistry department

Ph. Speeckaert
Member of the Chemical Engineering section of the Chemistry department

Centre d'Etude de l'Energie Nucléaire (C.E.N.)

MOL (Belgium)

An important part of this work has been performed in the scope of the joint Euratom/United States research and development program: Contracts 008-60-4/RDB, 049-61-4/RDB, 098-62-11/RDB.

Since the last Geneva Conference, the technique of volatilization reprocessing has changed: in many cases it has called for gas-solid reactions. Besides, the types of fuel to be taken into consideration are no longer the same : the regeneration of elements based on naturel uranium is not envisaged any more, whereas the ceramics are the object of intensive research.

Translation provided by courtesy of the submitting Government

The present document forms part of the contribution made by Belgium in this field. Different reprocessing cycles have been studied for some ceramic fuels in the form of oxide or carbide, and for some uranium alloys.

The oxides and the carbides are submitted to gas-solid reactions which, in one way, pulverize the fuel and facilitate the formation of plutonium and uranium hexafluorides, and, in the other way, permit the spreading of the heat liberated during these transformations.

In that way the sequences of the following reactions have been studied:

(1) $UO_2\ (PuO_2) \xrightarrow{O_2} U_3O_8\ (PuO_2) \xrightarrow{NH_3} UO_2(PuO_2) \longrightarrow UF_4(PuF_4) \xrightarrow{F_2} UF_6(PuF_6) \xrightarrow[\text{salts}]{\text{fused}} UO_2(PuO_2)$

(2) $UO_2\ (PuO_2) \xrightarrow{O_2} U_3O_8(PuO_2) \xrightarrow{HF} UF_4\text{-}UO_2F_2(PuF_4) \xrightarrow{F_2} UF_6(PuF_6) \xrightarrow[\text{salts}]{\text{fused}} UO_2(PuO_2)$

(3) $UO_2(PuO_2) \xrightarrow{O_2} U_3O_8(PuO_2) \xrightarrow{HF} UF_4\text{-}UO_2F_2(PuF_4) \xrightarrow{ClF_3} UF_6(PuF_4) \xrightarrow{F_2} (PuF_6) \xrightarrow[\text{salts}]{\text{fused}} (PuO_2)$

(4) $UO_2\ (PuO_2) \xrightarrow[O_2]{HNO_3} UO_3\ (PuO_2) \xrightarrow{SF_4} UF_6(PuF_4) \xrightarrow{F_2} (PuF_6) \xrightarrow[\text{salts}]{\text{fused}} (PuO_2)$

(5) $UC\ (PuC) \xrightarrow{HF} UF_4(PuF_4) \xrightarrow{ClF_3} UF_6(PuF_4) \xrightarrow{F_2} (PuF_6) \xrightarrow[\text{salts}]{\text{fused}} (PuO_2)$

(6) $UO_2 \xrightarrow[Cl_2]{CCl_4} UCl_6 \xrightarrow[H_2]{H_2O} U\ O_2$

Some of these cycles have been combined among themselves. The first and the fourth have been discarded, because of technological difficulties, as well as the long time required for the reactions, and their low yield. Preference has been given to cycles (2) and (3). These are characterized by the use of fluorine to effect the simultaneous volatilization of the uranium and the plutonium, or of chlorine trifluoride to obtain a successive volatilization of these two compounds; this successive volatilization appears particularly advantageous, because it permits the plutonium hexafluoride to be treated in an apparatus of small volume.

Although no operations have been performed on real fuels,

the results of pilot studies on cycles (2) and (3) give reason to believe that the recovery of uranium will surpass 99 per cent. A large technological effort has still to be made to apply the basic knowledge already acquired for the separation of uranium and plutonium and for the recuperation of the latter, in a suitable form.

Cycle (5) proposed for the reprocessing of carbides is a particularly interesting example of the great simplicity which the volatilization methods can attain.

The last cycle, schematised in the sequence of reactions (6), has stimulated an extensive study of the synthesis and properties of uranium chlorides and of those of fission products, as well as their complexes. Up to now it has not received concrete realizations.

The study of processing of alloys has pertained to the following elements: U-Zr, U-Mo, U-Nb, U-Al. Some chemical conditions of reaction and separation under the action of chlorinating agents have been studied. For the uranium-aluminium alloy, the research has evolved towards a cycle of processing by chlorination and fluorination, where hydrochloric acid is called upon to volatilize the aluminium; the residual uranium trichloride can subsequently be transformed into hexafluoride.

In the scope of the European economy, the processes proposed will be the most paying for fuels rich in fissile materials. Thus, volatilization should rather be considered as complementary to liquid-liquid extraction, of which the actual principal applications concern low enrichments. The numerous unsolved technological problems, as well as the necessity to verify the data acquired by preliminary work, cause a delay of 5 to 10 years to be foreseen before the birth of the first industrial applications.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/772
BELGIUM
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

THE USE OF INORGANIC ION EXCHANGERS IN ACID MEDIUM FOR THE RECOVERY OF CESIUM AND STRONTIUM FROM REPROCESSING SOLUTIONS.

L.Baetslé
Section Head

D.Van Deyck

D.Huys

A.Guery
Fellow on leave from Israeli Atomic Energy Commission, Israel

Waste Disposal Research Laboratory
Centre d'Etude de l'Energie Nucléaire (C.E.N.)
MOL (Belgium)

operated by the Société Belge de Chimie Nucléaire (BELCHIM)

Work done under contract C.E.N.-EURATOM N° 028-62-4/RISB.

The search for new synthetic mineral ion exchangers exhibiting selective uptake of long lived fission products has led to the elaboration of a separation scheme for cesium and strontium from reprocessing solutions.

Zirconium phosphate, whose chemical, physical and radiochemical properties are reported in connection with its structure shows selective ion exchange properties for cesium in acid medium. The loading of cesium reaches a maximum of 0.2 me/g when a solution containing 0.5 g/l cesium and 0.5 N in acidity is fed to the zirconium phosphate in the H^+ form. That corresponds to a concentration factor of about 60.

The results are presented for several types of reprocessing solutions a.o. those reported by the Hanford, Windscale and Marcoule plants and those expected at the Eurochemic plant.

A new family of synthetic ion exchangers was found by precipitating in acid medium Na_2MoO_4 or Na_2WO_4 solution with $H_4Fe(CN)_6$.

"The ferrocyanide molybdate ion exchanger" has an extremely high affinity for cesium which allows the separation of the latter nuclide from reprocessing waste solution of 2 M HNO_3. A concentration factor of nearly 100 may be achieved leading to a maximum loading of 0.5 me/g at this high acidity.

The difficulties encountered with the strontium separation in acid medium and in the presence of bulk ions are outlined but some promising results were obtained with clinoptilolite after dilution of the feed solution.

An experimental set-up built for the separation of strontium and cesium with mineral ion exchangers is reported with emphasis on its technology.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/773
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

SOME DEVELOPMENT WORK DONE AT EUROCHEMIC

by

E. Detilleux et al.

Eurochemic, Mol, Belgium [+]

The Eurochemic Research Department has been simultaneously working on the chemical and technological problems involved for the future operation of the Mol plant and several studies on new processes applicable in the field of aqueous reprocessing for irradiated nuclear fuels.

This paper reviews some aspects of the programme.

From the technological aspect, various studies on fuel dissolution, operation of extraction pulse columns and evaporation of radioactive solutions have been carried out. Two of these studies are discussed here in some detail : air pulsation of the extraction columns and interface detection by air bubbling.

The purpose of the work on air pulsation has been to increase the operational safety by elimination of the mechanical parts and a reduction in the maintenance requirements. The work on the interface control has shown that the air bubbling system, in continuous organic phase, is perfectly applicable to industrial columns.

[+] One of the joint undertakings of the O.E.C.D. European Nuclear Energy Agency.

Translation provided by courtesy of the submitting Government

From the chemical aspect, and in parallel to problems directly connected to the plant erection, the use of tertiary amines and quaternary ammonium salts as extraction media, and the dissolution of uranium and thorium carbide - graphite fuels, are more specially considered.

In the field of tertiary amines, a process for plutonium final purification has been developed. It involves extraction of the plutonium by trilaurylamine (TLA) followed by a precipitation, directly from the organic phase, of the plutonium as an oxalate.

The possibility of applying TLA (or other tertiary amines) to achieve, without any reducing agent, the separation of plutonium from uranium is also discussed. This process would also allow the partition of uranium and plutonium and the final purification of the latter to be combined in one cycle.

The substitution of the tertiary amines by quaternary ammonium salt, in particular for plutonium purification, is also discussed. They have the advantage of giving stable complexes with Pu(IV) (soluble in organic solvents) but allow easy back-extraction of the plutonium by using aqueous solutions either slightly acid, neutral or even alkaline.

The studies connected with the processing of graphite fuels have been mainly concerned with elements having uranium or thorium carbide fuels incorporated in complex carbide matrices.

Dry combustion and attack by a molten salt medium have been selected as a first possible step for the reprocessing.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/774
BELGIUM
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

CONDITIONING BY DISPERSION IN ASPHALT AND GROUND DISPOSAL OF MEDIUM ACTIVE CONCENTRATES

P. Dejonghe
Head, Waste Disposal Department

L. Baetslé
Section Head

N. Van de Voorde

W. Maes

P. Staner

J. Pyck

J. Souffriau

Waste Disposal Research Laboratory
Centre d'Etude de l'Energie Nucléaire (C.E.N.)
MOL (Belgium)

operated by the Société Belge de Chimie Nucléaire (BELCHIM)

Work performed in the scope of contracts C.E.N.-EURATOM N° 048-61-4/RDB/1.4.2 and 002-63-10/WAS B

The final storage of medium active waste concentrates has been studied under two essential aspects, the migration and dispersion of radioelements into the soil and also the conditioning or insolubilization of wastes before storage.

Migration and dispersion of radioelements in sandy soils :

The study of migration of radioelements in sandy soils required the development of both reproducible and accurate sampling techniques. These techniques are based on ion exchange or on imbibition on hygroscopic materials.

Laboratory studies on physico-chemical characteristics of soil materials led to a preliminary evaluation of the sorption capacity of the soil for radionuclides.

As a next step Sr^{89} and Cs^{137} were injected into the ground-water table and, using the above mentioned equipment, the evolution of the radioactive cloud has been determined.

The results of these experiments presented under the form of hydrodynamic and physico-chemical dispersion coefficients confirm to a large extent the laboratory studies both for migration rate and for the shape of the radioactive cloud.

The migration of radionuclides in a non-saturated profile is a much more complicated phenomenon. It has been demonstrated that there exists a gradual transition from plug flow to mixing flow.

The existence of preferential pathways became evident.

Finally the use of glauconite, lignite or phosphatic solutions as a barrier against radioelement migration has been discussed.
The application of such a technique improves the storage capacity of a site but represents also a supplementary protection around particularly hazardous nuclear facilities.

Insolubilization prior to storage :

Treatment concentrates, both liquid and solid, can be insolubilized by violent stirring in asphalt at a temperature of 200°C. This process completely dries the active material and produces after cooling a homogeneous mixture, practically insoluble in any natural medium. The dry matter content may amount up to 50 % of the mixture.

The pilot installation has an evaporation capacity of 25 litres water per hour and has successfully been operated during 18 months, both for insolubilization of semi-solids (65 % of moisture) and for the evaporation of liquid sludge or various solutions.

The radioactivity of the off gases , after purification by an electrofilter, represents about 10^{-5} of the input activity. Insolubilized blocks

of 200 litres, stored in the ground-water table at pH 4, showed no alteration after one year.
The radioactivity eluted during this period corresponds to a diffusion coefficient $D = 3.10^{-9} cm^2/day$.

The working expenses have been reported in detail and represent about 5 % of the total cost of the water treatment.

Owing to the results of these studies we have elaborated a storage program for radioactive concentrates at Mol located in a sandy profile and an area with a relatively dense population.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/775
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

SOME ASPECTS OF RADIOACTIVE WASTE DISPOSAL IN CZECHOSLOVAKIA AS A DENSELY POPULATED CONTINENTAL COUNTRY

L. Berák

The problems of decontamination and waste disposal are being investigated, in the Institute of Nuclear Research, both on laboratory and plant scale. In the field of the investigation of natural sorbents the following rocks seem to satisfy best the criteria for the selection of suitable column materials: zeolitically changed pumice ryolite as well as basalt and tefrite tuffs for Cs sorption, the best results for Sr sorption under certain conditions were shown by ryodacite tuff.

Since the degree of toxicity of radioactive waters is given by biologically hazardous radionuclides, especially Sr^{90}, ground disposal of radioactive waters containing Sr^{90} in concentrations exceeding by several orders the MPC values cannot be realized in densely populated countries. The maximum possibility of extraction of such a radionuclide seems to be provided by a substance capable of retaining Sr^{+} with highest selectivity, i.e. in a medium of Ca^{2+} hardness. The results of this investigation justify the opinion that the application of synthetized sorbents will lead to a marked progress in the decontamination of low- and medium-level wastes.

In the field of decontamination of waters containing anionic radioactivities, hydrated oxides showed high capacities and higher K_d values for S^{35} and P^{32} sorption than ionic precipitates.

Another developed method is based on the principle of an electrolytical dissolution of Al or Fe electrodes, when in the presence of phosphate ions in the solution flakes of Al or Fe phosphates are formed. This results in an effective separation of alkaline earth elements, radiocolloids and colloids from the solution. The device proposed the use of two stages,

primary classical chemical precipitation in an excess of phosphate, and sorption on a precipitate of aluminium and iron phosphate formed by anodic dissolution.

In an advanced stage of development is a complex method of a counter-current sorption on sorbent on suspension, which is governed by laws of extraction and is effective for the separation of not only ions but also radiocolloids. The technological apparatus consists of 3 stages. As the final decontamination stage, columns filled with a cheap natural sorbent are being developed.

In the study of insolubilisation of radioisotopes, great attention has been paid to the behaviour of the solid phase in the thermal treatment of melted rocks. In the investigation of the diffusion of Sr and Cs in melted basalt, changes of the activation energy of diffusion with temperature have been observed. Above the recrystallisation point, a sudden increase of the activation energy takes place, which confirms the hypothesis on the relative immobilization of fixed ions in the recrystallized phase.

In order to enable the operation of the Institute, two decontamination units of radioactive wastes have been built. The first unit is equipped with two types of evaporators, one for a capacity of 200 l/hr and the other for 130 l/hr. The first evaporator is a classical boiler type and the second one a film type, both with duplicator heating.

For the full operation of the Institute, especially of the radiochemical laboratories, a large unit of an output of several m^3/hr has been built. The unit is equipped with a classical decontamination device. The major part of waste waters is treated by chemical precipitation. For wastes with higher radioactivities, an evaporator of an output of $2m^3$/hr is installed in the unit. The sludge from the chemical precipitation is filtered by means of a rotation filter and fixed into cement. The volume of solid wastes is reduced by pressing.

In view of the unfavourable geological conditions of the territory of CSSR for direct ground disposal, the technological practice is directed to old underground spaces selected by suitable geological and hydrogeological conditions.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/776
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

REACTIVITY OPTIMUM COMPENSATION IN A MULTIZONE REACTOR FROM THE POINT OF VIEW OF MAXIMUM POWER PLANT OUTPUT

J. Schmid and V. Stach

Nuclear power plant reactors have a relatively large reactivity excess during the operation with a continuous refuelling, especially with fresh fuel elements. To compensate this reactivity excess, a number of absorption rods affecting considerably the reactor power must be used. The aim of the present paper is to determine the conditions for the minimization of the power plant output losses, i.e. to determine the sequence and insertion depth of individual compensating rods.

The calculations refer to a heavy water gas cooled reactor. This reactor type has a relatively small number of technology channels. In the paper it is assumed the compensating rods affect the four immediately neighboring channels only (square lattice) and the reactor as well as the absorption rod distance are so large that the interference of the inserted rods is negligible.

Basic relations for determining the thermal output of the given technology channel as the functions of the insertion depth of the neighboring absorption rod are given. The problem is solved both for the case of a constant gas flow in the channel and for such a channel flow to retain the maximum permissible fuel element temperature. The second case is solved for two different conditions, i.e. for compensating rod insertion from the coolant inlet side and for the insertion from the coolant outlet side.

Further the paper gives the relations for the determination of the compensating rods' effect on thermal and electrical reactor output for both controlled and uncontrolled gas flow in individual channels.

The last part of the paper deals with the problem of optimum compensation of the given reactivity excess. Some functions are introduced to characterize the relative electric output loss referred to the relative absorption rod efficiency. According to the course of these functions both optimum insertion depth of the absorption rod and optimum rod insertion sequence in a multizone reactor may be determined.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/777
DENMARK
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Design Problems in the D_2O-Moderated, Organic-Cooled Reactor Concept DOR

by

M. Møller-Madsen and F. List

The Danish Atomic Energy Commission
Research Establishment Risö, Roskilde, Denmark

and

J. Elming

Burmeister and Wain's Maskin- og Skibsbyggeri A/S
(Engineers and Shipbuilders)
Copenhagen, Denmark

A feasibility study of a heavy-water-moderated, organic-cooled power reactor called DOR is being carried out by the Danish Atomic Energy Commission at the Research Establishment Risö.

The aims of the feasibility study are summarized in the paper, and a description of the mechanical design of DOR is given. The heavy-water moderator is contained in a calandria tank, and hot pressure tubes containing fuel elements and coolant run vertically through the calandria tank. As fuel, UO_2 and UC clad with SAP are considered. SAP material is also used for the coolant pressure tubes through the core. Stainless steel is used for the calandria tank, which has zircaloy tubes rolled into the top and the bottom tube plate. DOR is designed for an output of about 250 MWe.

The mechanical design of coolant tubes, fuel elements and calandria tank is considered to present some of the major problems in the DOR project; therefore, efforts have largely been concentrated on the development of these components. Under research and development contracts, Danish industrial

companies are participating in this development work. In the paper the present design of the components mentioned above is discussed.

A survey of the experimental work which has been carried out is given. The experiments are related to coolant channels and calandria tank and comprise experiments with rolled joints between SAP and stainless steel and between zircaloy and stainless steel. Further, experimental investigations of the stiffness of calandria tube plates and of coolant pressure drop across the core are described.

Finally an account is given of the experimental work which is at the preparatory stage. These experiments are concerned with an out-of-pile loop with a DOR coolant channel in full size, an out-of-pile facility for the investigation of heat transmission with electrically heated dummy fuel elements, an organic in-pile loop, and test sections of the calandria tank.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/778
GERMANY, FEDERAL REPUBLIC OF
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Potentialities of heavy-water moderated gas cooled pressure tube reactors

by

W. Keller

Siemens-Schuckertwerke AG

Big heavy-water moderated power reactors are, if properly designed, the best thermal converters on the uranium fuel-cycle, and potentially near breeders on the thorium fuel-cycle. Amongst several possible coolants for such a reactor, CO_2 shows various important advantages: First, it is the low neutron absorption of the coolant, which allows, in connection with a low absorbing fuel canning material, a fuel cycle with natural or slightly enriched uranium. With natural uranium in form of UO_2 fuel burn-ups of 8000 MWd/t should be achievable. Secondly, CO_2 is compatible with many structural materials at high temperatures. These make it possible to obtain steam conditions similar to those of modern conventional steam power stations. Thirdly, the D_2O-CO_2 pressure tube reactor has the advantage that the heavy-water is kept at a low temperature and nearly at atmospheric pressure. The risk of heavy water losses is therefore kept very low, especially because loading and shuffling of fuel under power can be accomplished without disturbing the integrity of the heavy-water circuit. Fourthly, as the UO_2 is compatible with CO_2, small canning failures will not lead to serious consequences. With UO_2 and very low enrichment there is every hope for reaching high burn-ups in the range

of 20 000 MWd/t, and low fuel cycle costs. In addition to these advantages CO_2 at high pressure (80 - 120 ata) permits power densities comparable to those in water cooled D_2O-cores.

Four years of development at Siemens-Schuckertwerke AG in connection with a design contract by Atomkraft Bayern for a CO_2-D_2O prototype plant of 100 MW_e, lead us to the conviction that the potentialities of this reactor-type can be realized in the near future. At 550°C reactor outlet temperature the use of low alloyed steel for the primary circuit seems feasible; while as canning material of the fuel rods very thin stainless steel may be used, special zirconium alloys show even more favourable prospects, beryllium, besides this, not being out of the picture.

Extrapolations to larger plants (> 300 MW_{el}) are leading to construction costs, which are as low as in any other heavy-water reactor concept as yet proposed. The familiarity of CO_2-technology should permit scaling up to plants of larger size at an early date.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/779
GERMANY, FEDERAL REPUBLIC OF
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Objectives in the Development of D_2O Pressure Vessel Reactors

by

A. Ziegler

Siemens-Schuckertwerke AG

The fundamental reason for the development of heavy water moderated reactors for power production is their superior neutron economy. The advantages gained thereby must, from the point of view of economy, balance the higher costs incurred. While the question whether this is possible had been discussed at numerous conferences the final answer has so far not been given. The experiences gained by us in the construction of the multipurpose reactor MZFR are apt to confirm the widely-held expectation that D_2O moderated reactors are able to compete successfully with other types. In this paper, three important aspects of the problem are discussed. On the one hand it is examined how much lower the fuel costs are than in the case of enriched fuel. On the other hand the capital investment for D_2O and a fuel load are compared with the cost of a core load for light water reactors. Thirdly the author discusses with regard to larger units, how much their investment and operating costs are likely to exceed those of light water reactors.

Hitherto, the main obstacle for the further development was generally seen in the limitation of power imposed by the restricted size of the pressure vessel. The paper examines how far the

upper limit for the power can presumably be raised by developing a pressure vessel made of pre-stressed concrete. By enlarging the vessel it is possible to reduce the neutron leakage to such an extent that, with U-233-Th as a fuel, a conversion factor above 1 can be obtained. The breeding gain will at the least be sufficient to maintain by reprocessing, after a burn-up of about 30000 mwd/t, a constant reactivity of the reactor. The breeding cycle can also be initiated with U-235 or technical Pu. It offers, under certain conditions, economic aspects for the utilisation of Pu as favourable as those of a fast breeder fuelled with oxide. Looking beyond the short-range objectives the author discusses how far the investment costs may be reduced by the admission of boiling with subsequent nuclear superheating, in conjunction with a direct steam cycle feeding the turbine. As a first step in this direction the MZFR ist now being equipped with loops for boiling and superheating experiments.